APPENDIX E
EVALUATION OF HUMAN HEALTH EFFECTS FROM NORMAL OPERATIONS

E.1 INTRODUCTION

This appendix provides a brief general discussion on radiation and its associated health effects and describes the method and assumptions used for estimating the potential impacts and risks to individuals and the general public from exposure to the releases of radioactivity and hazardous chemicals during normal operations at the proposed facilities. Information regarding potential radiological impacts resulting from facility accidents is provided in Appendix F of this environmental impact statement (EIS).

This appendix presents numerical information using engineering and/or scientific notation. For example, the number 100,000 also can be expressed as 1×10^5. The fraction 0.00001 also can be expressed as 1×10^{-5}. The following chart defines the equivalent numerical notations that may be used in this appendix.

FRACTIONS AND MULTIPLES OF UNITS

Multiple	*Decimal Equivalent*	*Prefix*	*Symbol*
1×10^6	1,000,000	mega-	M
1×10^3	1,000	kilo-	k
1×10^2	100	hecto-	h
1×10	10	deka-	da
1×10^{-1}	0.1	deci-	d
1×10^{-2}	0.01	centi-	c
1×10^{-3}	0.001	milli-	m
1×10^{-6}	0.000001	micro-	μ
1×10^{-9}	0.000000001	nano-	n
1×10^{-12}	0.000000000001	pico-	p
1×10^{-15}	0.000000000000001	femto-	f
1×10^{-18}	0.000000000000000001	atto-	a

E.2 RADIOLOGICAL IMPACTS ON HUMAN HEALTH

Radiation exposure and its consequences are topics of interest to the general public. For this reason, this EIS places much emphasis on the consequences of exposure to radiation, provides the reader with background information on the nature of radiation, and explains the basic concepts used in the evaluation of radiation health effects.

E.2.1 Nature of Radiation and Its Effects on Humans

What Is Radiation?

Radiation is energy transferred in the form of particles or waves. Globally, human beings are exposed constantly to radiation from the solar system and the earth's rocks and soil. This radiation contributes to the natural background radiation that always surrounds us. Manmade sources of radiation also exist, including medical and dental x-rays, household smoke detectors, and materials released from nuclear and coal-fired power plants.

All matter in the universe is composed of atoms. Radiation comes from the activity of tiny particles within an atom. An atom consists of a positively charged nucleus (central part of an atom) with a number of negatively charged electron particles in various orbits around the nucleus. There are two types of particles in the nucleus: neutrons that are electrically neutral and protons that are positively charged. Atoms of different types are known as elements. There are more than 100 natural and manmade elements. An element has equal numbers of electrons and protons. When atoms of an element differ in their number of neutrons, they are called isotopes of that element. All elements have three or more isotopes, some or all of which could be unstable (i.e., decay with time).

Unstable isotopes undergo spontaneous change, known as radioactive disintegration or radioactive decay. The process of continuously undergoing spontaneous disintegration is called radioactivity. The radioactivity of a material decreases with time. The time it takes a material to lose half of its original radioactivity is its half-life. An isotope's half-life is a measure of its decay rate. For example, an isotope with a half-life of eight days will lose one-half of its radioactivity in that amount of time. In eight more days, one-half of the remaining radioactivity will be lost, and so on. Each radioactive element has a characteristic half-life. The half-lives of various radioactive elements may vary from millionths of a second to millions of years.

As unstable isotopes change into more stable forms, they emit electrically charged particles. These particles may be either an alpha particle (a helium nucleus) or a beta particle (an electron), with various levels of kinetic energy. Sometimes these particles are emitted in conjunction with gamma rays. The alpha and beta particles are frequently referred to as ionizing radiation. Ionizing radiation refers to the fact that the charged particle energy force can ionize, or electrically charge, an atom by stripping off one of its electrons. Gamma rays, even though they do not carry an electric charge as they pass through an element, can ionize its atoms by ejecting electrons. Thus, they cause ionization indirectly. Ionizing radiation can cause a change in the chemical composition of many things, including living tissue (organs), which can affect the way they function.

When a radioactive isotope of an element emits a particle, it changes to an entirely different element, one that may or may not be radioactive. Eventually a stable element is formed. This transformation, which may take several steps, is known as a decay chain. For example, radium, which is a member of the radioactive decay chain of uranium, has a half-life of 1,622 years. It emits an alpha particle and becomes radon, a radioactive gas with a half-life of only 3.8 days. Radon decays first to polonium, then through a series of further decay steps to bismuth, and ultimately to lead, which is a stable element. Meanwhile, the decay products will build up and eventually die away as time progresses.

The characteristics of various forms of ionizing radiation are briefly described below and in the box at right (see Chapter 6 for further definition):

Radiation Type	Typical Travel Distance in Air	Barrier
α	Couple of centimeters	Sheet of paper or skin's surface
β	Few meters	Thin sheet of aluminum foil or glass
γ	Very large [a]	Thick wall of concrete, lead, or steel
n	Very large	Water, paraffin, graphite

[a] *Would be infinite in a vacuum*

Alpha (α)

Alpha particles are the heaviest type of ionizing radiation. They can travel only a couple of centimeters in air. Alpha particles lose their energy almost as soon as they collide with anything. They can be stopped easily by a sheet of paper or by the skin's surface.

Beta (β)

Beta particles are much (7,330 times) lighter than alpha particles. They can travel a longer distance than alpha particles in the air. A high-energy beta particle can travel a few meters in the air. Beta particles can pass through a sheet of paper, but may be stopped by a thin sheet of aluminum foil or glass.

Gamma (γ)

Gamma rays (and x-rays), unlike alpha or beta particles, are waves of pure energy. Gamma rays travel at the speed of light. Gamma radiation is very penetrating and requires a thick wall of concrete, lead, or steel to stop it.

Neutrons (n)

Neutrons are particles that contribute to radiation exposure both directly and indirectly. The most prolific source of neutrons is a nuclear reactor. Indirect radiation exposure occurs when gamma rays and alpha particles are emitted following neutron capture in matter. A neutron has about one-quarter the weight of an alpha particle. It will travel in the air until it is absorbed in another element.

Units of Radiation Measure

During the early days of radiological experience, there was no precise unit of radiation measure. Therefore, a variety of units were used to measure radiation. These units were used to determine the amount, type, and intensity of radiation. Just as heat can be measured in terms of its intensity or effects using units of calories or degrees, amounts of radiation or its effects can be measured in units of curies, radiation absorbed dose (rad), or dose equivalent (roentgen equivalent man, or rem). The following summarizes those units (see also the definitions in the Glossary [Chapter 6]).

Curie

The curie, named after the French scientists Marie and Pierre Curie, describes the "intensity" of a sample of radioactive material. The rate of decay of 1 gram of radium is the basis of this unit of measure. It is equal to 3.7×10^{10} disintegrations (decays) per second.

Rad

The rad is the unit of measurement for the physical absorption of radiation. The total energy absorbed per unit quantity of tissue is referred to as absorbed dose (or simply dose). As sunlight heats pavement by giving up an amount of energy to it, radiation similarly gives up rads of energy to objects in its path. One rad is equal to the amount of radiation that leads to the deposition of 0.01 joule of energy per kilogram of absorbing material.

Radiation Units and Conversions to International System of Units

1 curie = 3.7×10^{10} becquerel
1 rad = 0.01 gray
1 rem = 0.01 sievert
1 gray = 1 joule per kilogram
1 becquerel = 1 disintegration per second

Rem

A rem is a measurement of the dose equivalent from radiation based on its biological effects. The rem is used in measuring the effects of radiation on the body as degrees centigrade are used in measuring the effects of sunlight heating pavement. Thus, 1 rem of one type of radiation is presumed to have the same biological effects as 1 rem of any other kind of radiation. This allows comparison of the biological effects of radionuclides that emit different types of radiation.

The units of radiation measure in the International System of Units are: becquerel (a measure of source intensity [activity]), gray (a measure of absorbed dose), and sievert (a measure of dose equivalent).

An individual may be exposed to ionizing radiation externally (from a radioactive source outside the body) or internally (from ingesting or inhaling radioactive material). The external dose is different from the internal dose because an external dose is delivered only during the actual time of exposure to the external radiation source, but an internal dose continues to be delivered as long as the radioactive source is in the body. The dose from internal exposure is calculated over 50 years following the initial exposure. Both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time.

Sources of Radiation

The average American receives a total of approximately 360 millirem per year from all sources of radiation, both natural and manmade, of which approximately 300 millirem per year are from natural sources. The sources of radiation can be divided into six different categories: (1) cosmic radiation, (2) terrestrial radiation, (3) internal radiation, (4) consumer products, (5) medical diagnosis and therapy, and (6) other sources (NCRP 1987). These categories are discussed in the following paragraphs.

Cosmic Radiation

Cosmic radiation is ionizing radiation resulting from energetic charged particles from space continuously hitting the earth's atmosphere. These particles and the secondary particles and photons they create comprise cosmic radiation. Because the atmosphere provides some shielding against cosmic radiation, the intensity of this radiation increases with the altitude above sea level. The average dose to people in the United States from this source is approximately 27 millirem per year.

External Terrestrial Radiation

External terrestrial radiation is the radiation emitted from the radioactive materials in the Earth's rocks and soils. The average dose from external terrestrial radiation is approximately 28 millirem per year.

Internal Radiation

Internal radiation results from the human body metabolizing natural radioactive material that has entered the body by inhalation or ingestion. Natural radionuclides in the body include isotopes of uranium, thorium, radium, radon, polonium, bismuth, potassium, rubidium, and carbon. The major contributor to the annual dose equivalent for internal radioactivity is the short-lived decay products of radon, which contribute approximately 200 millirem per year. The average dose from other internal radionuclides is approximately 39 millirem per year.

Consumer Products

Consumer products also contain sources of ionizing radiation. In some products, such as smoke detectors and airport x-ray machines, the radiation source is essential to the product's operation. In other products, such as televisions and tobacco, the radiation occurs as the product's function. The average dose from consumer products is approximately 10 millirem per year.

Medical Diagnosis and Therapy

Radiation is an important diagnostic medical tool and cancer treatment. Diagnostic x-rays result in an average exposure of 39 millirem per year. Nuclear medical procedures result in an average exposure of 14 millirem per year.

Other Sources

There are a few additional sources of radiation that contribute minor doses to individuals in the United States. The dose from nuclear fuel cycle facilities (e.g., uranium mines, mills, and fuel processing plants), nuclear power plants, and transportation routes has been estimated to be less than 1 millirem per year. Radioactive fallout from atmospheric atomic bomb tests, emissions of radioactive material from nuclear facilities, emissions from certain mineral extraction facilities, and transportation of radioactive materials contribute less than 1 millirem per year to the average dose to an individual. Air travel contributes approximately 1 millirem per year to the average dose.

Exposure Pathways

As stated earlier, an individual may be exposed to ionizing radiation both externally and internally. The different ways that could result in radiation exposure to an individual are called exposure pathways. Each type of exposure is discussed separately in the following paragraphs.

External Exposure

External exposure can result from several different pathways, all having in common the fact that the radiation causing the exposure is external to the body. These pathways include exposure to a cloud of radiation passing over the receptor (i.e., an individual member of the public), standing on ground that is contaminated with radioactivity, and swimming or boating in contaminated water. If the receptor departs from the source of radiation exposure, the dose rate will be reduced. It is assumed that external exposure occurs uniformly during the year. The appropriate dose measure is called the effective dose equivalent.

Internal Exposure

Internal exposure results from a radiation source entering the human body through either inhalation of contaminated air or ingestion of contaminated food or water. In contrast to external exposure, once a

radiation source enters the body, it remains there for a period of time that varies depending on decay and biological half-life. The absorbed dose to each organ of the body is calculated for a period of 50 years following the intake. The calculated absorbed dose is called the committed dose equivalent. Various organs have different susceptibilities to harm from radiation. The quantity that takes these different susceptibilities into account is called the committed effective dose equivalent, and it provides a broad indicator of the risk to the health of an individual from radiation. The committed effective dose equivalent is a weighted sum of the committed dose equivalent in each major organ or tissue. The concept of committed effective dose equivalent applies only to internal pathways.

Radiation Protection Guides

Various organizations have issued radiation protection guides. The responsibilities of the main radiation safety organizations, particularly those that affect policies in the United States, are summarized below.

International Commission on Radiological Protection

This Commission has the responsibility for providing guidance in matters of radiation safety. The operating policy of this organization is to prepare recommendations to deal with basic principles of radiation protection and to leave to the various national protection committees the responsibility of introducing the detailed technical regulations, recommendations, or codes of practice best suited to the needs of their countries.

National Council on Radiation Protection and Measurements

In the United States, this Council is the national organization that has the responsibility for adapting and providing detailed technical guidelines for implementing the International Commission on Radiological Protection recommendations. The Council consists of technical experts who are specialists in radiation protection and scientists who are experts in disciplines that form the basis for radiation protection.

National Research Council/National Academy of Sciences

The National Research Council is an organization within the National Academy of Sciences that associates the broad community of science and technology with the Academy's purposes of furthering knowledge and advising the Federal Government.

Environmental Protection Agency

The Environmental Protection Agency (EPA) has published a series of documents, *Radiation Protection Guidance to Federal Agencies*. This guidance is used as a regulatory benchmark by a number of Federal agencies, including the U.S. Department of Energy (DOE), in the realm of limiting public and occupational work force exposures to the greatest extent possible.

Limits of Radiation Exposure

Limits of exposure to members of the public and radiation workers are derived from International Commission on Radiological Protection recommendations. The EPA utilizes the National Commission on Radiological Protection and the International Commission on Radiological Protection recommendations and sets specific annual exposure limits (usually less than those specified by the Commission) in Radiation Protection Guidance to Federal Agency documents. Each regulatory organization then establishes its own set of radiation standards. DOE has established a set of limits for radiation workers in 10 CFR 835. **Table E–1** provides the various exposure limits set by DOE and the EPA for radiation workers and members of the public.

Table E–1 Exposure Limits for Members of the Public and Radiation Workers

Guidance Criteria (Organization)	*Public Exposure Limits at the Site Boundary*	*Worker Exposure Limits*
40 CFR 190 (EPA)	25 millirem per year (all pathways)	—
10 CFR 835 (DOE)	—	5,000 millirem per year [a]
DOE Order N441.1 (DOE)	—	2,000 millirem per year [a]
DOE Order 5400.5 (DOE) [b]	10 millirem per year (all air pathways) 4 millirem per year (drinking water pathway) 100 millirem per year (all pathways)	—
40 CFR 61 (EPA)	10 millirem per year (all air pathways)	—
40 CFR 141 (EPA)	4 millirem per year (drinking water pathways)	—

[a] Although these are limits (or levels) which are enforced by DOE, worker doses must still adhere to as low as reasonably achievable principles.

[b] Derived from 40 CFR 61, 40 CFR 141, and 10 CFR 20.

E.2.2 Health Effects

Radiation exposure and its consequences are topics of interest to the general public. To provide the background for discussions of impacts, this section explains the basic concepts used in the evaluation of radiation effects.

Radiation can cause a variety of damaging health effects in people. The most significant effects are induced cancer fatalities. These effects are referred to as "latent" cancer fatalities because the cancer may take many years to develop. In the discussions that follow, all fatal cancers are considered latent; therefore, the term "latent" is not used.

The National Research Council's Committee on the Biological Effects of Ionizing Radiation (BEIR) has prepared a series of reports to advise the U.S. Government on the health consequences of radiation exposures. *Health Effects of Exposure to Low Levels of Ionizing Radiation*, BEIR V (National Research Council 1990), provides the most current estimates for excess mortality from leukemia and other cancers that are expected to result from exposure to ionizing radiation. BEIR V provides estimates that are consistently higher than those in its predecessor, BEIR III. This increase is attributed to several factors, including the use of a linear dose response model for cancers other than leukemia, revised dosimetry for the Japanese atomic bomb survivors, and additional follow-up studies of the atomic bomb survivors and associated others. BEIR III employs constant, relative, and absolute risk models, with separate coefficients for each of several sex and age-at-exposure groups. BEIR V develops models in which the excess relative risk is expressed as a function of age at exposure, time after exposure, and sex for each of several cancer categories. The BEIR III models were based on the assumption that absolute risks are comparable between the atomic bomb survivors and the U.S. population. BEIR V models were based on the assumption that the relative risks are comparable. For a disease such as lung cancer, where baseline risks in the United States are much larger than those in Japan, the BEIR V approach leads to larger risk estimates than the BEIR III approach.

The models and risk coefficients in BEIR V were derived through analyses of relevant epidemiologic data that included the Japanese atomic bomb survivors, ankylosis spondylitis patients, Canadian and Massachusetts fluoroscopy (breast cancer) patients, New York postpartum mastitis (breast cancer) patients, Israeli tinea capitis (thyroid cancer) patients, and Rochester thymus (thyroid cancer) patients. Models for leukemia, respiratory cancer, digestive cancer, and other cancers used only the atomic bomb survivor data, although results of analyses of the ankylosis spondylitis patients were considered. Atomic bomb survivor analyses were based on revised dosimetry, with an assumed relative biological effectiveness of 20 for neutrons, and were restricted to doses less than 400 rads. Estimates of risks of fatal cancers, other than leukemia, were obtained by totaling the estimates for breast cancer, respiratory cancer, digestive cancer, and other cancers.

The National Council on Radiation Protection and Measurements (NCRP 1993), based on the radiation risk estimates provided in BEIR V and the International Commission on Radiological Protection Publication 60 recommendations (ICRP 1991), has estimated the total detriment resulting from low dose[1] or low dose rate exposure to ionizing radiation to be 0.00073 per rem for the general population and 0.00056 per rem for the working population. The total detriment includes fatal and nonfatal cancer and severe hereditary (genetic) effects. The major contribution to the total detriment is from fatal cancer and is estimated to be 0.0004 and 0.0005 per rem for the radiation workers and the general population, respectively. **Table E–2** provides the breakdown of the risk factors for both workers and the general population. Nonfatal cancers and genetic effects are less probable consequences of radiation exposure. To simplify the presentation of the impacts, estimated effects of radiation are calculated only in terms of latent cancer fatalities.

Table E–2 Nominal Health Effects Coefficients (Risk Factors) From Exposure to 1 Rem of Ionizing Radiation

Exposed Individual	*Fatal Cancer* [a, c]	*Nonfatal Cancer* [b]	*Genetic Disorders* [b]	*Total*
Worker	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] For fatal cancer, the health effect coefficient is the same as the probability coefficient. When applied to an individual, the units are the lifetime probability of a latent cancer fatality per rem of radiation dose. When applied to a population of individuals, the units are the excess number of cancers per person-rem of radiation dose.

[b] In determining a means of assessing health effects from radiation exposure, the International Commission on Radiological Protection has developed a weighting method for nonfatal cancers and genetic effects. Genetic effects can be applied only to a population, not individuals.

[c] For high individual exposures (greater than or equal to 20 rem), the health factors are multiplied by a factor of 2.

Source: NCRP 1993.

The numerical estimates of fatal cancers presented in this EIS were obtained using a linear extrapolation from the nominal risk estimated for lifetime total cancer mortality, which is 0.1 gray (10 rad). Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of fatal cancers. Studies of human populations exposed to low doses are inadequate to demonstrate the actual level of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiologic observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

Health Effect Risk Factors Used in This EIS

Health impacts from radiation exposure, whether from external or internal sources, generally are identified as "somatic" (i.e., affecting the exposed individual) or "genetic" (i.e., affecting descendants of the exposed individual). Radiation is more likely to produce somatic effects than genetic effects. The somatic risks of most importance are induced cancers. Except for leukemia, which can have an induction period (time between exposure to carcinogen and cancer diagnosis) of as little as 2 to 7 years, most cancers have an induction period of more than 20 years.

For a uniform irradiation of the body, the incidence of cancer varies among organs and tissues; the thyroid and skin demonstrate a greater sensitivity than other organs. Such cancers, however, also produce relatively low mortality rates because they are relatively amenable to medical treatment. Because fatal cancer is the

[1] *Low dose is defined as the dose level where DNA repair can occur in a few hours after irradiation-induced damage. Currently, a dose level of about 0.2 grays (20 rad), or a dose rate of 0.1 milligrays (0.01 rad) per minute is considered to allow the DNA to repair itself in a short period (EPA 1994).*

most probable serious effect of environmental and occupational radiation exposures, estimates of cancer fatalities rather than cancer incidence are presented in this EIS. The numbers of fatal cancers can be used to compare the risks among the various alternatives.

Based on the preceding discussion and the values presented in Table E–2, the fatal cancers to the general public during normal operations and for accidents in which individual doses are less than 20 rem are calculated using a health risk factor of 0.0005 per person-rem. For workers, a risk factor of 0.0004 excess fatal cancers per person-rem is used. (The risk factors are lifetime probabilities that an individual would develop a latent fatal cancer per rem of radiation.) This lower value reflects the absence of children (who are more radiosensitive than adults) in the work force. Nonfatal cancer and genetic disorders among the public are 20 and 26 percent, respectively, of the fatal cancer risk factor. For workers, the health risk estimators are both 20 percent of the fatal cancer risk factor. The nonfatal cancer risk factors are not used in this EIS.

The fatal cancer factors are used to calculate the statistical expectation of the effects of exposing a population to radiation. For example, if 100,000 people were each exposed to one time radiation dose of 100 millirem (0.1 rem), the collective dose would be 10,000 person-rem. The exposed population would then be expected to experience 5 additional latent cancer fatalities from the radiation (10,000 person-rem × 0.0005 lifetime probability of latent cancer fatalities per person-rem = 5 latent cancer fatalities).

Calculations of the number of excess fatal cancers associated with radiation exposure do not always yield whole numbers; calculations may yield numbers less than 1.0, especially in environmental impact applications. For example, if a population of 100,000 were exposed to a total dose of only 0.001 rem per person, the collective dose would be 100 person-rem, and the corresponding estimated number of latent cancer fatalities would be 0.05 (100,000 persons × 0.001 rem × 0.0005 latent cancer fatalities per person-rem = 0.05 latent cancer fatalities). The 0.05 means that there is one chance in 20 that the exposed population would experience one latent fatal cancer. In other words, the 0.05 latent cancer fatalities is the *expected* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people. In most groups, no person (0 people) would incur a latent fatal cancer from the 0.001 rem dose each member would have received. In a small fraction of the groups, 1 latent cancer fatality would result; in exceptionally few groups, 2 or more latent cancer fatalities would occur. The *average* expected number of deaths over all the groups would be 0.05 latent cancer fatalities (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is 0 latent cancer fatalities.

The same concept is applied to estimate the effects of radiation exposure on an individual member of the public. Consider the effects of individual's exposure to a 360 millirem (0.36 rem) annual dose from all radiation sources. The probability that the individual will develop a latent fatal cancer from continuous exposure to this radiation over an average life of 72 years (presumed) is 0.013 (1 person × 0.36 rem per year × 72 years × 0.0005 latent cancer fatality risk per person rem = 0.013). This correlates to one chance in 77 that the individual would develop a fatal cancer.

E.3 METHODOLOGY FOR ESTIMATING RADIOLOGICAL IMPACTS

The radiological impacts from releases during normal operation of the facilities used to treat and manage sodium-bonded spent nuclear fuel were calculated using Version 1.485 of the GENII computer code (PNL 1988). Site-specific input data were used including location, meteorology, population, and source terms. Section E.3.1 briefly describes GENII and outlines the approach used for normal operations.

E.3.1 GENII Computer Code

The GENII computer model, developed by Pacific Northwest National Laboratory, is an integrated system of various computer modules that analyze environmental contamination resulting from acute or chronic

releases to, or initial contamination in, air, water, or soil. The model calculates radiation doses to individuals and populations. The GENII computer model is well documented for assumptions, technical approach, method, and quality assurance issues. The GENII computer model has gone through extensive quality assurance and quality control steps, including comparing results from model computations with those from hand calculations and performing internal and external peer reviews (PNL 1988).

GENII code consists of several modules for various applications, see the code manual (PNL 1988) for details. For this EIS, only the ENVIN, ENV, and DOSE computer modules were used. The output of one module is stored in a file that can be used by the next module in the system. The functions of the three GENII computer modules used in this EIS are discussed below.

ENVIN

The ENVIN module of the GENII code controls the reading of input files and organizes the input for optimal use in the environmental transport and exposure module, ENV. The ENVIN code interprets the basic input, reads the basic GENII data libraries and other optional input files, and organizes the input into sequential segments based on radionuclide decay chains.

A standardized file that contains scenario, control, and inventory parameters is used as input to ENVIN. Radionuclide inventories can be entered as functions of releases to air or water, concentrations in basic environmental media (air, soil, or water), or concentrations in foods. If certain atmospheric dispersion options have been selected, this module would generate tables of atmospheric dispersion parameters that are used in later calculations. If the finite plume air submersion option is selected in addition to the atmospheric dispersion calculations, preliminary energy-dependent finite plume dose factors can be prepared as well. The ENVIN module prepares the data transfer files that are used as input by the ENV module; ENVIN generates the first portion of the calculation documentation—the run input parameters report.

ENV

The ENV module calculates the environmental transfer, uptake, and human exposure to radionuclides that result from the chosen scenario for the user-specified source term. The code reads the input files from ENVIN and then, for each radionuclide chain, sequentially performs the precalculations to establish the conditions at the start of the exposure scenario. Environmental concentrations of radionuclides are established at the beginning of the scenario by assuming decay of pre-existing sources, considering biotic transport of existing subsurface contamination, and defining soil contamination from continuing atmospheric or irrigation depositions. For each year of postulated exposure, the code then estimates the air, surface soil, deep soil, groundwater, and surface water concentrations of each radionuclide in the chain. Human exposures and intakes of each radionuclide are calculated for: (1) pathways of external exposure from finite atmospheric plumes; (2) inhalation; (3) external exposure from contaminated soil, sediments, and water; (4) external exposure from special geometries; and (5) internal exposures from consumption of terrestrial foods, aquatic foods, drinking water, animal products, and inadvertent intake of soil. The intermediate information on annual media concentrations and intake rates are written to data transfer files. Although these may be accessed directly, they are usually used as input to the DOSE module of GENII.

DOSE

The DOSE module reads the intake and exposure rates defined by the ENV module and converts the data to radiation dose.

E.3.2 Data and General Assumptions

To perform the dose assessments for this EIS, different types of data were collected and generated. This section discusses the various data, along with the assumptions made for performing the dose assessments in this EIS.

Dose assessments were performed for both members of the general public and workers around and at Argonne National Laboratory-West (ANL-W) and the Savannah River Site (SRS). These assessments were made to determine the incremental doses that would be associated with the alternatives addressed in this EIS. Incremental doses for members of the public were calculated (via GENII) for two different types of receptors:

- Maximally Exposed Offsite Individual—The maximally exposed offsite individual was assumed to be an individual member of the public located at a position on the site boundary that would yield the highest impacts during normal operations.

- Population—The general population living within 80 kilometers (50 miles) of the facility.

Meteorological Data

The meteorological data used for all normal operational scenarios discussed in this EIS were in the form of joint frequency data files. A joint frequency data file is a table listing the fractions of time the wind blows in a certain direction, at a certain speed, and within a certain stability class. The joint frequency data files were based on measurements taken over a period of several years at both the ANL-W and SRS sites.

Population Data

Population distributions were based on the 1990 Census of Population and Housing data (DOC 1992). Projections were determined for the year 2010 (representative year for operations) for areas within 80 kilometers (50 miles) of the release locations at ANL-W and SRS. The projected site-specific population in 2010, assumed to be representative of the population over the operational period evaluated, was used in the impact assessments. The population was spatially distributed on a circular grid with 16 directions and 10 radial distances up to 80 kilometers (50 miles). The grid was centered at the location from which the radionuclides were assumed to be released.

Source Term Data

The site- and process-specific source terms used to calculate the impacts of normal operations are provided in Section E.4.

Food Production and Consumption Data

Generic food consumption rates are established in the U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.109 (NRC 1977). This regulatory guide provides guidance for evaluating ingestion doses from consuming contaminated terrestrial and animal food products using a standard set of assumptions for crop and livestock growth and harvesting characteristics. In this EIS, food consumption rates were based on site-specific agricultural production rates and local diets.

Basic Assumptions

To estimate annual radiological impacts from normal operations, the following additional assumptions and factors were considered in using GENII:

- Radiological airborne gaseous and particulate emissions were assumed to be released to the atmosphere through the plant stacks. See Section E.4 for the specifics at each management facility.

- Ground contamination was based on dry deposition of radionuclides from normal operation releases, assuming no previously deposited radionuclides. Doses resulting from previously deposited radionuclides are accounted for in the baseline dose analysis, as presented in Chapter 3, and are not attributable to the processing of sodium-bonded spent nuclear fuel.

- Unless limited by the process duration, the inhalation exposure time to the plume was assumed to be per year for the maximally exposed offsite individual and the general population. Plume exposure parameters used in the GENII model for normal operations are provided in **Table E–3**.

- The exposed individual or population was assumed to have the characteristics and habits (e.g., inhalation and ingestion rates) of an adult human.

- A semi-infinite/finite plume model was used for air immersion doses. Other pathways evaluated were ground exposure, inhalation, and ingestion of food crops and animal products contaminated by deposition of radioactivity from the air.

- Resuspension of particulates was not considered because calculations of dust loading in the atmosphere show that this pathway is negligible compared to the other pathways.

- Reported release heights were used for atmospheric releases and were assumed to be the effective stack heights. The resultant doses were conservative, as use of the actual stack heights negates plume rise.

- The calculated doses were 50-year committed doses from 1 year of intake.

- Unless otherwise noted, radionuclide materials were considered to be released in the chemical form resulting in the largest radiological impact, thus maximizing the potential dose effect.

Table E–3 GENII Exposure Parameters to Plumes and Soil Contamination (Normal Operations)

Maximally Exposed Offsite Individual				*General Population*			
External Exposure		*Inhalation of Plume*		*External Exposure*		*Inhalation of Plume*	
Plume (hours)	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cubic centimeters per second)*	*Plume (hours)*	*Soil Contamination (hours)*	*Exposure Time (hours)*	*Breathing Rate (cubic centimeters per second)*
6,136	6,136	8,766	270	4,383	4,383	8,766	270

Sources: PNL 1988, NRC 1977.

Worker doses associated with the processing alternatives were determined from historical data associated with similar operations. See Section E.4 for details.

E.3.3 Uncertainties

The sequence of analyses performed to generate the radiological impact estimates from normal operation include: (1) selection of normal operational modes, (2) estimation of source terms, (3) estimation of environmental transport and uptake of radionuclides, (4) calculation of radiation doses to exposed individuals, and (5) estimation of health effects. There are uncertainties associated with each of these steps. Uncertainties exist in the way the physical systems being analyzed are represented by the computational models and in the data required to exercise the models (due to measurement, sampling, or natural variability).

In principle, one can estimate the uncertainty associated with each source and predict the remaining uncertainty in the results of each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final results. However, conducting such a full-scale quantitative uncertainty analysis is neither practical nor a standard practice for a study of this type. Instead, the analysis is designed to ensure—through judicious selection of release scenarios, models, and parameters—that the results represent the potential risks. This is accomplished by making conservative assumptions in the calculations at each step. The models, parameters, and release scenarios used in the calculations are selected in such a way that most intermediate results and, consequently, the final estimates of impacts are greater than would be expected. As a result, even though the range of uncertainty in a quantity might be large, the value calculated for the quantity would be close to one of the extremes in the range of possible values, so the chance of the actual quantity being greater than the calculated value would be low (or the chance of the quantity being less than the calculated value if the criteria are such that the quantity has to be maximized). The goal of the radiological assessment for normal operation in this study is to produce results that are conservative.

The degree of conservatism in the calculated results is related closely to the range of possible values the quantity can have. This range is determined by what can be expected to occur realistically. Limitations on the processing of material (e.g., design capacity/processing rate, system availability, operational duration) provide upper limits to the quantity of spent nuclear fuel that can be processed in a given time, (e.g., annually). In many cases these restrictions were used to represent normal operating capacity, thus maximizing the amount of spent nuclear fuel that can be processed annually. Using these upper limits on processing rates provides a conservative estimate of the annual release of radionuclides during normal operation for each of the treatment techniques. These conservative release estimates were used to calculate the annual impacts presented for each alternative.

Details of some of the proposed treatment processes (e.g., melt and dilute) have not been finalized, yet the evaluation of worker doses can be performed using data associated with existing operations, where appropriate. While this introduces additional uncertainties in the estimation of worker exposures, many similarities between existing and proposed operations justify the use of this data. Among the features that justify use of existing data are the following: ANL-W and SRS are both committed to adhering to as low as reasonably achievable radiation protection practices; both sites have treated spent nuclear fuel under similar operating conditions; existing facilities (although modified in some cases) and existing protective features will be used; and any operational controls generated for new processes will be similar to existing operational (procedural) controls. These similarities between existing and proposed process controls mitigate some of the uncertainties inherent in estimating the impacts of processes yet to be finalized.

The radionuclide composition of source terms has been estimated conservatively. There are uncertainties in the radionuclide inventory which are proportional to the quantities of source terms that ultimately are released. For evaluation purposes, the inventory used is based on the spent nuclear fuel with the highest representative radionuclide content with no credit taken for further decay beyond that which occurred prior to the year 2000.

E.4 RADIOLOGICAL RELEASES TO THE ENVIRONMENT AND ASSOCIATED IMPACTS

This section summarizes the estimated radiological releases to the environment as well as resulting impacts associated with the various alternatives assessed in this EIS. Impacts to workers from these alternatives also are discussed. The methodology for estimating radiological impacts, associated input data, and analytical assumptions is provided in Section E.3.

E.4.1 Electrometallurgically Treat Blanket and Driver Fuel at ANL-W (Alternative 1)

Under this alternative, releases of radioactive material would occur during normal operational processing of the sodium-bonded fuel rods in the argon cell at the Fuel Conditioning Facility. Fuel assemblies would be disassembled in the Fuel Conditioning Facility air cell, and individual fuel elements then would be transferred to the argon cell for chopping and treatment in one of the electrorefiners. The entire inventory of gaseous fission products, mainly tritium and krypton-85, is assumed to be released during processing in the Fuel Conditioning Facility. The likelihood of release of radionuclides other than the gaseous fission products is very small. No radionuclides would be released from the packaged salt and packaged metallic waste material transferred from the Fuel Conditioning Facility to the Hot Fuel Examination Facility.

Estimated radioactive releases during normal operations at ANL-W were calculated using a conservative methodology. First, assumptions were made to estimate a maximum annual throughput of material to be processed at the Fuel Conditioning Facility. There would be two electrorefiners in the Fuel Conditioning Facility argon cell; blanket material would be treated in one of the two electrorefiners and driver material would be treated in the other. Both driver and blanket material could be processed each year. Based on an annual operational processing limit of 5,000 kilograms (11,023 pounds) of total heavy metal fuel material consisting of more than 600 kilograms (1,320 pounds) of heavy metal driver material, it was assumed that driver fuel would be processed at the maximum rate until all driver fuel was processed. In addition, it was assumed that the Experimental Breeder Reactor-II (EBR-II) fuel (driver and blanket) currently at ANL-W would be processed first. Using these assumptions, annual mass processing throughputs were developed for the purposes of estimating releases of radioactive material during normal operations, and are presented in **Table E–4**.

Radioactive releases from the Fuel Conditioning Facility argon cell during fuel treatment were estimated next. Radioactivity associated with the fuel to be processed was determined using the fuel radioactivity inventory values discussed in Appendix D. Estimated releases were based on a methodology developed in support of ANL-W's State of Idaho and National Emission Standards for Hazardous Air Pollutants air permitting activities, and agreed upon by the State of Idaho's Department of Environmental Quality (Bauer 1992). From this methodology, equilibrium concentrations in the argon cell (curies per cubic meter per curie processed) were calculated and applied to the inventory associated with the assumed annual throughputs shown in Table E–4. Annual radioactive releases to the atmosphere were calculated as the product of the radionuclide equilibrium concentrations in the argon cell, the annual argon cell atmosphere exhaust (74,400 cubic meters per year), and a conservative adjustment (0.00001) to account for the combined filtration of the two banks of high-efficiency particulate air filters that the cell exhaust must pass through before entering the environment. This filtration adjustment was not applied to tritium or krypton-85, as 100 percent of these radionuclides were assumed to be released.

The Fuel Conditioning Facility stack was modeled with an effective stack height of 60.96 meters (200 feet). This is the actual stack height, and for conservatism, no plume rise was included in the atmospheric dispersion modeling.

Table E–4 Annual Processing Assumptions for Estimation of Radiological Releases During Normal Operations Under Alternative 1 at ANL-W

Year of Processing	Driver Fuel (kilograms per year)		Blanket Fuel (kilograms per year)		Total Fuel (kilograms per year)		
	EBR-II [a]	Fast Flux Test Facility [b]	EBR-II [c]	Fermi-1	Driver	Blanket	Driver + Blanket
1	600	0	4,400	0	600	4,400	5,000
2	600	0	4,400	0	600	4,400	5,000
3	600	0	4,400	0	600	4,400	5,000
4	600	0	4,400	0	600	4,400	5,000
5	600	0	4,400	0	600	4,400	5,000
6	100	400	400	4,200	500	4,600	5,100
7	0	0	0	5,000	0	5,000	5,000
8	0	0	0	5,000	0	5,000	5,000
9	0	0	0	5,000	0	5,000	5,000
10	0	0	0	5,000	0	5,000	5,000
11	0	0	0	5,000	0	5,000	5,000
12	0	0	0	5,000	0	5,000	5,000
Totals (kilograms)	3,100	400	22,400	34,200	3,500	56,600	60,100

[a] EBR-II driver spent nuclear fuel consists of 1,100 kilograms of EBR-II driver spent nuclear fuel at ANL-W and 2,000 kilograms at INTEC.

[b] The Fast Flux Test Facility driver spent nuclear fuel consists of 250 kilograms of sodium-bonded Fast Flux Test Facility driver spent nuclear fuel at Hanford, 70 kilograms of unirradiated sodium-bonded Fast Flux Test Facility fuel, and 80 kilograms of miscellaneous spent nuclear fuel at INTEC, Sandia National Laboratory, SRS, and the Oak Ridge Reservation.

[c] EBR-II blanket spent nuclear fuel consists of EBR-II blanket spent nuclear fuel at ANL-W.

The dose resulting from the release of tritium (H-3) depends heavily on its chemical form. The inhalation dose from oxidized tritium is 25,000 times higher than for tritium in elemental form (ICRP 1982). The dose conversion factors used in the GENII code assume that tritium released to the environment is in the oxidized form and therefore are very conservative for releases that involve elemental tritium. Because of the argon atmosphere in the Fuel Conditioning Facility argon cell, releases of tritium to the cell atmosphere would not become oxidized, and stack releases of tritium most likely would be in the elemental form. The oxidation of elemental tritium to oxidized tritium has been shown to occur slowly in the environment, and for this EIS, the long-term dose from elemental tritium releases is conservatively estimated to be 1 percent of that for the oxidized form (DOE 1997). Therefore, the inventory of tritium for each year of electrometallurgical treatment processing at the Fuel Conditioning Facility was multiplied by a factor of 0.01 to convert them to an equivalent release of tritium oxide for use as input to the GENII code.

Radiological Gaseous Emissions

The estimated annual and total atmospheric releases are tabulated in **Table E–5**. This table lists only those radionuclides that resulted from a screening procedure to indicate potential significant dose contributions. The source term listed in Table E–5 for each of the first five years of processing (years 1 through 5) represents the source term that results in the highest annual offsite dose, and is therefore used for the maximum annual dose calculations. The project lifetime total values in Table E–5 represent the total estimated releases over the 12 years of processing at ANL-W.

Table E–5 Annual and Total Radiological Releases During Normal Operations Under Alternative 1 at ANL-W

Isotope [a]	Annual Releases (curies per year)			Project Lifetime Total (curies)
	Years 1 through 5	Year 6	Years 7 through 12	
H-3	770	680	0.38	4,530
C-14	1.7×10^{-12}	1.0×10^{-12}	2.3×10^{-16}	9.4×10^{-12}
Fe-55	1.4×10^{-8}	1.5×10^{-8}	5.8×10^{-13}	8.7×10^{-8}
Co-60	1.6×10^{-9}	9.7×10^{-10}	1.9×10^{-12}	8.8×10^{-9}
Ni-63	6.5×10^{-10}	1.7×10^{-10}	1.0×10^{-12}	3.4×10^{-9}
Kr-85	11,570	8,800	3.3	66,670
Sr-90	7.0×10^{-8}	5.2×10^{-8}	4.7×10^{-11}	4.0×10^{-7}
Y-90	7.0×10^{-8}	5.2×10^{-8}	4.7×10^{-11}	4.0×10^{-7}
Ru-106	3.2×10^{-8}	2.9×10^{-8}	7.6×10^{-17}	1.9×10^{-7}
Rh-106	3.2×10^{-8}	2.9×10^{-8}	7.6×10^{-17}	1.9×10^{-7}
Cd-113m	6.7×10^{-10}	5.2×10^{-10}	3.1×10^{-13}	3.9×10^{-9}
Sb-125	4.1×10^{-8}	3.6×10^{-8}	3.2×10^{-13}	2.4×10^{-7}
Te-125m	4.5×10^{-10}	3.9×10^{-10}	3.4×10^{-15}	2.6×10^{-9}
I-129	1.4×10^{-12}	9.7×10^{-13}	1.8×10^{-15}	8.2×10^{-12}
Cs-134	3.2×10^{-8}	4.0×10^{-8}	9.5×10^{-16}	2.0×10^{-7}
Cs-137	4.0×10^{-6}	2.9×10^{-6}	3.5×10^{-9}	0.000023
Ba-137m	3.8×10^{-6}	2.8×10^{-6}	3.3×10^{-9}	0.000022
Ce-144	1.2×10^{-9}	1.8×10^{-9}	1.9×10^{-20}	7.7×10^{-9}
Pr-144	1.2×10^{-9}	1.8×10^{-9}	1.9×10^{-20}	7.7×10^{-9}
Pm-147	2.9×10^{-8}	2.6×10^{-8}	2.3×10^{-13}	1.7×10^{-7}
Sm-151	2.1×10^{-9}	1.4×10^{-9}	3.7×10^{-12}	1.2×10^{-8}
Eu-154	2.1×10^{-10}	2.0×10^{-10}	2.2×10^{-15}	1.3×10^{-9}
Eu-155	1.4×10^{-9}	1.1×10^{-9}	1.9×10^{-13}	8.3×10^{-9}
Th-228	1.6×10^{-14}	1.3×10^{-14}	3.2×10^{-19}	9.1×10^{-14}
U-234	1.2×10^{-11}	7.8×10^{-12}	7.8×10^{-17}	6.7×10^{-11}
U-235	3.9×10^{-13}	2.6×10^{-13}	1.8×10^{-14}	2.3×10^{-12}
U-236	3.7×10^{-13}	2.6×10^{-13}	2.7×10^{-16}	2.1×10^{-12}
U-238	7.4×10^{-13}	7.7×10^{-13}	8.1×10^{-13}	9.4×10^{-12}
Np-237	3.9×10^{-13}	2.8×10^{-13}	2.1×10^{-15}	2.2×10^{-12}
Pu-238	2.9×10^{-10}	2.2×10^{-10}	3.4×10^{-14}	1.6×10^{-9}
Pu-239	7.1×10^{-9}	1.2×10^{-9}	1.4×10^{-10}	3.7×10^{-8}
Pu-240	4.7×10^{-10}	1.2×10^{-10}	1.1×10^{-13}	2.5×10^{-9}
Pu-241	1.9×10^{-9}	1.1×10^{-9}	3.6×10^{-15}	1.1×10^{-8}
Am-241	6.2×10^{-12}	1.8×10^{-12}	1.5×10^{-17}	3.3×10^{-11}
Am-242m	6.4×10^{-14}	9.3×10^{-15}	3.4×10^{-23}	3.3×10^{-13}
Totals	12,310	9,500	3.7	71,200

[a] The listed isotopes are present within the argon cell at the Fuel Conditioning Facility. Due to lack (scarcity) of oxygen in the argon cell, the tritium (H-3) released to the cell would be in molecular (elemental) form.

Population Impacts

The estimated annual radiological impacts due to the source term for the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are tabulated in **Table E–6**. Calculated impacts are shown for each year of processing as well as for each of the fuel types to be processed. Impacts are listed resulting from releases during processing EBR-II driver and blanket spent nuclear fuel during each of the first five years (years 1 through 5), processing some of all four fuel types during the sixth year (year 6), and processing Fermi-1 blanket spent nuclear fuel during each of the final six years (years 7 through 12). The impacts to the maximally exposed offsite individual and the surrounding population would result primarily from estimated releases of tritium (H-3) and krypton-85. Together, these two radionuclides would account for greater than 99.9 percent of the estimated impacts.

Table E–6 Annual Radiological Impacts to the Public From Operational Activities Under Alternative 1 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	***Spent Nuclear Fuel Type***	***Collective Dose (person-rem per year)***	***Latent Cancer Fatalities (number of cancers)***	***Dose (millirem per year)***	***Latent Cancer Fatality Risk***
1 - 5	EBR-II driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	Fast Flux Test Facility driver	0	0	0	0
	EBR-II blanket	0.000083	4.2×10^{-8}	0.000010	5.0×10^{-12}
	Fermi-1 blanket	0	0	0	0
	All fuel, years 1 through 5	0.0028	1.4×10^{-6}	0.00034	1.7×10^{-10}
6	EBR-II driver	0.00046	2.3×10^{-7}	0.000054	2.7×10^{-11}
	Fast Flux Test Facility driver	0.0018	9.2×10^{-7}	0.00022	1.1×10^{-10}
	EBR-II blanket	7.6×10^{-6}	3.8×10^{-9}	9.1×10^{-7}	4.6×10^{-13}
	Fermi-1 blanket	9.1×10^{-7}	4.5×10^{-10}	1.1×10^{-7}	5.5×10^{-14}
	All fuel, year 6	0.0023	1.2×10^{-6}	0.00028	1.4×10^{-10}
7 - 12	EBR-II driver	0	0	0	0
	Fast Flux Test Facility driver	0	0	0	0
	EBR-II blanket	0	0	0	0
	Fermi-1 blanket	1.1×10^{-6}	5.4×10^{-10}	1.3×10^{-7}	6.5×10^{-14}
	All fuel, years 7 through 12	1.1×10^{-6}	5.4×10^{-10}	1.3×10^{-7}	6.5×10^{-14}

Total cumulative radiological impacts over the projected 13 years of operations under this alternative are tabulated in **Table E–7**. This table shows the sum of the calculated impacts to the maximally exposed offsite individual and the surrounding population over 12 years of fuel treatment.

Table E–7 Cumulative Maximum Radiological Impacts to the Public From Normal Operational Releases Under Alternative 1 at ANL-W

	Population		*Maximally Exposed Offsite Individual*	
	Collective Dose (person-rem)	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem)*	*Latent Cancer Fatality Risk*
Project total impacts [a]	0.0163	8.2×10^{-6}	0.00198	9.9×10^{-10}

[a] Total impacts are estimated for the 12-year duration of fuel treatment; there are no releases in the 13th year, i.e., only salt stabilization is performed.

Worker Impacts

Workers involved with electrometallurgical treatment activities at ANL-W could receive radiation doses during handling activities, such as receiving and unloading fuel casks, and transferring in-process waste material from the Fuel Conditioning Facility to the Hot Fuel Examination Facility. Doses received during in-cell activities likely would be very small. A maximally exposed worker dose estimate for this EIS is based on the regulatory limit of 5,000 millirem per year for radiation workers at DOE sites. If an individual worker received this dose each year of the 13 years of the electrometallurgical treatment project, the total worker dose would be 65,000 millirem with an associated risk of developing fatal cancer of 0.026.

However, actual worker doses are likely to be much lower than this maximum estimate. The ANL-W radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. In addition, ANL-W has established an administrative goal of 1,500 millirem per year to any individual. The general design goals at the Fuel Conditioning Facility, for example, were to maintain radiation fields below 0.5 millirem per hour at all workstations. This means that for an individual working at the Fuel Conditioning Facility for a full-time occupational work year of 2,000 hours, the annual dose would be 1,000 millirem.

Worker population doses were estimated by examining the type and duration of various operations performed by workers involved with the electrometallurgical treatment project. Doses can be estimated based on previous doses from similar activities at ANL-W. Based on information from ANL-W, the total worker population dose estimate is 22 person-rem per year, averaging out to an individual dose of 60 millirem per year for each of the 346 involved workers. If these estimates are extended out over the 13 years of operational activities (12 years of fuel treatment and a year of high-level radioactive waste conversion activities), the collective worker dose is 286 person-rem and the associated risk is 0.11 latent cancer fatalities. The estimated impacts to the worker population associated with this alternative are summarized in **Table E–8**.

Table E–8 Annual and Total Impacts to Workers From Operational Activities Under Alternative 1 at ANL-W

	Worker Population	
	Collective Dose (person-rem)	*Latent Cancer Fatalities*
Annual impacts	22	0.0088
Project total impacts [a]	319	0.13

[a] Total impacts are estimated for the 13-year processing duration, plus a year for deactivation activities at 33 person-rem.

E.4.2 Prepare Blanket Fuel and Electrometallurgically Treat Driver Fuel at ANL-W (Alternatives 2 Through 5)

In Alternatives 2 through 5, the blanket spent nuclear fuel assemblies would need to be prepared at the ANL-W facilities prior to packaging in high-integrity cans or processing in either the plutonium-uranium extraction (PUREX) process at SRS or the melt and dilute process at SRS or ANL-W. When the blanket spent nuclear fuel is to be processed at SRS, Alternative 3 (PUREX processing) and 5 (melt and dilute processing), the blanket spent nuclear fuel would be declad and cleaned at ANL-W in the argon cell of the Hot Fuel Examination Facility. Processing of the blanket spent nuclear fuel assemblies at ANL-W (Alternative 2, placing the blanket spent nuclear fuel in high-integrity cans, and Alternatives 4 and 6, melt and dilute) would not require decladding of the blanket spent nuclear fuel. This activity also would be performed in the argon cell of the Hot Fuel Examination Facility. The preparation of the blanket spent nuclear fuel under these alternatives would require only that the fuel be cut into segments and cleaned (see Appendix C for details). The following discussion addresses the radiological impact of normal operations at ANL-W for the preparation of the blanket spent nuclear fuel elements and the electrometallurgical treatment of the driver spent nuclear fuel elements. This analysis is applicable to Alternatives 2 through 5.

Gaseous Emissions

Blanket spent nuclear fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated in Section E.4.1, krypton-85 and elemental tritium are the most prevalent gaseous radionuclides that would be released to the environment. The released tritium (H-3) into the cell would not be oxidized because of a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Appendix E, Section E.4.1, provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium. The maximum released curies of radioactive gaseous emissions occurs when preparation of the blanket spent nuclear fuel and chopping of the driver spent nuclear fuel (for electrometallurgical treatment processing) are performed simultaneously. This simultaneous operation was estimated to occur over a six-year period starting in 2003. Based on a blanket spent nuclear fuel preparation throughput of 10 metric tons of heavy metal and an electrometallurgical treatment process rate of about 0.6 metric tons of heavy metal of driver spent nuclear fuel elements annually, at most about 809 curies of elemental tritium and 11,860 curies of krypton-85 would be released to the atmosphere annually; see **Table E–9**. This release rate would last about two years, or until all of the EBR-II blanket spent nuclear fuel is processed; then the release rate would drop during the processing of the Fermi-1 blanket spent nuclear fuel (the release rate for the processing of 10 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85). The gaseous fission products generated during treatment processes at the Hot Fuel Examination Facility would be released to the atmosphere through the 31-meter (94-foot) facility stack.

Table E–9 Maximum Annual Radiological Gaseous Emission From Activities Associated With Alternatives 2 Through 5 at ANL-W

Spent Nuclear Fuel Type	*Facility*	*Maximum Processing Rate (metric tons of heavy metal per year)*	*Duration (years)*	*Annual Release (curies)*	
				Tritium [a]	*Krypton-85*
Driver fuel	Fuel Conditioning Facility	0.6	6	738	11,340
EBR-II blanket fuel	Hot Fuel Examination Facility	10	2.4	71.2	520
Fermi-1 blanket fuel	Hot Fuel Examination Facility	10	3.6	0.76	6.6
Maximum annual release [b]			2.4	809	11,860

[a] Elemental tritium; about 1 percent of tritium was assumed to be in oxidized form. See discussion in Section E.4.1.
[b] Maximum annual release occurs during concurrent processing of EBR-II driver and blanket spent nuclear fuel at ANL-W.

Population Impacts

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are presented in **Table E–10**. As stated in Section E.4.1 the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E–10 result from releases that are assumed to be 1 percent oxidized tritium, the same assumption used in the analysis of Alternative 1. These impacts are calculated for the preparation of the blanket spent nuclear fuel assemblies, for the processing of the driver spent nuclear fuel assemblies using the electrometallurgical treatment process, and the maximum total impacts. The maximum annual impact is associated with the concurrent treatment of EBR-II driver and blanket spent nuclear fuel. Under alternatives 2 through 5, treatment of the driver and blanket spent nuclear fuel does not begin at the same time. Electrometallurgical treatment of the driver spent nuclear fuel is expected to begin three years before the beginning of treatment of the blanket spent nuclear fuel (see the discussion for each alternative in Chapter 4). This results in the nine-year treatment duration identified in Table E–10, where only driver spent nuclear fuel is treated in the first three years; both driver and blanket spent nuclear fuel are treated in years four through six; and only blanket spent nuclear fuel is treated in the final three years. In Alternative 4, which includes melt and dilute processing of blanket spent nuclear fuel at ANL-W, the gaseous radionuclides, which result in over 99 percent of the offsite dose (tritium, krypton, and iodine), are released during the declad and clean process. It is the six-year duration of this portion of the melt and dilute process that was used as the time frame for modeling the operation releases from the treatment of blanket spent nuclear fuel for Alternative 4. As stated earlier, treatment of Fermi-1 spent nuclear fuel at ANL-W would have a negligible contributing impact. These impacts are applicable to the processing of blanket and driver spent nuclear fuel under Alternatives 2, 3, 4, and 5 at ANL-W.

Table E–10 Annual and Total Radiological Impacts to the Public From Normal Operational Releases Under Alternatives 2 Through 5 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	*Spent Nuclear Fuel Type*	*Collective Dose (person-rem per year)*	*Latent Cancer Fatalities (number of cancers)*	*Dose (millirem per year)*	*Latent Cancer Fatality Risk*
1-3	Driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	Blanket	0	0	0	0
	All fuel	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
4-5	Driver	0.0027	1.4×10^{-6}	0.00033	1.6×10^{-10}
	EBR-II blanket	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
	All fuel	0.0030	1.5×10^{-6}	0.00038	1.9×10^{-10}
6	Driver	0.0023	1.2×10^{-6}	0.00028	1.4×10^{-10}
	EBR-II blanket	0.00011	5.6×10^{-8}	0.000019	9.6×10^{-12}
	Fermi-1 blanket	1.9×10^{-6}	9.7×10^{-10}	3.3×10^{-7}	1.6×10^{-13}
	All fuel	0.0024	1.2×10^{-6}	0.00030	1.7×10^{-10}
7-9	Driver	0	0	0	0
	Fermi-1 blanket	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
	All fuel	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
Project total [a]		0.0165	8.3×10^{-6}	0.0021	1.0×10^{-9}

[a] Maximum annual radiological impacts occur during two years of concurrent EBR-II driver and blanket spent nuclear fuel processing.

Worker Impacts

The worker activities under Alternatives 2 through 5 at ANL-W would be similar to those under Alternative 1. Therefore, the annual worker dose and the worker population dose would be similar to those provided in Section E.4.1. The project total dose is provided in Section 4.4.4.1.

E.4.3 PUREX Processing at SRS (Alternative 3)

PUREX processing at F-Canyon would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket spent nuclear fuel. Since declad and cleaned blanket spent nuclear fuel would be packaged and sent to SRS, no additional gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that the gaseous fission products in the blanket spent nuclear fuel would remain within the fuel matrix and would be released to the environment (from the facility stack, 60 meters [198 feet] high) during PUREX processing at SRS. As a result, there would be incurred doses to the public associated with PUREX operations. The duration of PUREX operations was estimated to be six months, based on the F-Canyon's throughput and consistent with assumptions made for the treatment duration of a similar-type fuel at SRS in the *SRS Spent Nuclear Fuel Management Final Environmental Impact Statement* (DOE 2000).

Gaseous Emissions

According to SRS Spent Nuclear Fuel EIS data (DOE 1997), tritium (H-3) and krypton-85 are the only isotopes that would be expected to be released during PUREX processing operations. Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed that the inventory of krypton-85 and tritium would be released. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket spent nuclear fuel, the potential airborne radiological release quantities were estimated and are presented in **Table E–11**. This inventory was used to calculate the population doses from air emissions.

Table E–11 Estimated Incremental Releases of Radiological Air Emissions and Liquid Effluent During Normal Operations of PUREX Processing Under Alternative 3 at SRS

Isotope	*Releases to Air (curies)*	*Releases to Liquid (curies)* [a]
H-3	162	1.54
Kr-85	1,188	-
Sr-89/90	-	0.000031
Cs-137	-	0.0022
U-234	-	0.000085
U-235	-	0.000011
U-238	-	0.00019
Pu-238	-	0.000016
Pu-239	-	7.76×10^{-6}

[a] Estimated curies using the information provided in the SRS Environmental Data for 1997 (Arnett and Mamatey 1998).

Liquid Effluent

PUREX processing is the only process among the alternatives considered that would release measurable radioactive nuclides to the surface water. This release would occur through the cooling water system. The expected radiological effluent from processing declad and cleaned blanket spent nuclear fuel at F-Canyon were estimated based on the measured data from various effluent streams at F-Area, as presented in the SRS

Environmental Report and Data for 1997 (Arnett and Mamatey 1998). Since the mechanism associated with releases of liquid effluent from PUREX processing at F-Canyon is essentially the same for almost every fuel type processed, the F-Area 1997 effluent data were used to conservatively represent the potential releases from a six-month operation of F-Canyon. Table E–11 lists the radionuclides and their corresponding curies that are estimated to be released during PUREX processing of blanket spent nuclear fuel.

Population Impacts

Estimated annual radiological impacts associated with the F-Canyon PUREX operations for the maximally exposed offsite individual and the general population residing within the 80-kilometer (50-mile) radius surrounding F-Canyon are presented in **Table E–12**. This table provides the radiological doses to the public from air emissions and liquid effluent separately. According to the SRS Environmental Report, a maximally exposed offsite individual associated with liquid releases is an individual who lives downriver of SRS 365 days per year, drinks 2 liters of untreated water per day from the Savannah River, consumes a large amount of Savannah River fish, and spends the majority of time on or near the river. The general population liquid effluent dose is calculated for the discrete population groups at Beaufort-Jasper and Port Wentworth, as well as for other diffuse population groups that make use of the Savannah River; the majority of this dose is due to the drinking water pathway.

For conservatism, as well as demonstrating compliance with DOE Order 5400.5 (100 millirem annual dose limit to an individual from all pathways), the incremental airborne and liquid doses associated with the F-Canyon processing were summed together even though two distinct individuals are assumed to receive a maximum airborne and a maximum liquid dose. In addition, for analysis purposes, it was assumed that tritium would be released to the atmosphere in oxide form. The public impacts from radiological liquid effluent were estimated based on the results provided in the SRS's Interim Management of Nuclear Materials EIS (DOE 1995). This is consistent with the approach used in the recent SRS Spent Nuclear Fuel Management Final EIS (DOE 2000), which used "per unit" values (per metric tons of fuel processed) to estimate liquid doses associated with the PUREX processing of 20 metric tons of heavy metal of declad blanket spent nuclear fuel. This EIS uses the same approach to estimate the radiological doses to the public from potential radiological liquid effluent from PUREX processing.

Table E–12 Annual and Total Radiological Impacts to the Public From Normal Operational Releases During PUREX Processing Under Alternative 3 at SRS

Population [a]				*Maximally Exposed Offsite Individual* [a]			
Air Dose (person-rem)	***Liquid Dose*** [b] ***(person-rem)***	***Total Collective Dose (person-rem)***	***Latent Cancer Fatalities***	***Air Dose (millirem)***	***Liquid Dose*** [b] ***(millirem)***	***Total Dose (millirem)***	***Latent Cancer Fatality Risk***
0.019	0.00068	0.020	0.000010	0.00039	0.00012	0.00051	2.6×10^{-10}

[a] The dose values presented apply to both annual and project total, since the processing is done in less than a year.
[b] The dose values were estimated based on the results for processing a similar fuel presented in the Interim Management of Nuclear Materials EIS (DOE 1995).

Worker Impacts

Worker population and worker doses associated with PUREX processing at SRS were based on 300 workers and the site administration dose limit of 500 millirem per year for each worker and are consistent with those presented in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a

six-month processing period. **Table E–13** presents estimated values to both the average worker and entire work force population.

Table E–13 Annual and Cumulative Worker Radiological Impacts from Normal Operational Activities Under Alternative 3 at SRS

Worker Population		*Individual Worker*	
Collective Dose (person-rem per year)	*Latent Cancer Fatalities From Six Months of Processing*	*Individual Dose (millirem per year)*	*Latent Cancer Fatality Risk From Six Months of Processing*
75 [a]	0.015	500 [a, b]	1.0×10^{-4}

[a] Processing of blanket spent nuclear fuel will require six months of F-Canyon operation, yielding half of the annual doses presented.
[b] 500 millirem is an annual ALARA administrative dose limit at SRS. The average worker dose is about 50 millirem (DOE 2000).

E.4.4 SRS Building 105-L Melt and Dilute Radiological Releases and Impacts (Alternative 5)

Melt and dilute processing at Building 105-L would release radioactive gaseous fission products during treatment of about 57 metric tons of heavy metal of EBR-II and Fermi-1 blanket spent nuclear fuel. Since declad and cleaned blanket spent nuclear fuel would be packaged and sent to SRS, no additional gaseous fission products would be expected to be present in that fuel. However, it was assumed conservatively that the gaseous fission products in the blanket spent nuclear fuel would remain within the fuel matrix and would be released to the environment from the facility stack (62 meters [203 feet] high) during melt and dilute processing at SRS. As a result, there would be incurred doses to the public associated with these operations. The duration of the melt and dilute process was estimated to be about three years, based on the current design throughput of the melter and an assumption that the final metallic high-level radioactive waste product from this process would contain about 30 percent depleted uranium in aluminum alloy (WSRC 1999).

Gaseous Emissions

Based on the assumption that the entire fission gas inventory would remain within the fuel matrix after the decladding and cleaning process, it was assumed the inventory of krypton-85 and tritium (H-3) would be released during the melt and dilute process. Using the gaseous fission product inventory provided in Appendix D for the EBR-II and Fermi-1 blanket spent nuclear fuel, the potential airborne radiological release quantities were estimated and are presented in **Table E–14**. These inventories then were used to calculate the population doses from air emissions.

Table E–14 Annual Radiological Releases During Normal Melt and Dilute Operations at Building 105-L Under Alternative 5 at SRS

Isotope	*Releases [a] to Air (curies)*
H-3	54
Kr-85	396

[a] There are no liquid releases associated with melt and dilute processing at SRS.

Liquid Effluent

The melt and dilute process would not produce liquid effluent.

Population Impacts

Estimated annual radiological impacts associated with melt and dilute operations at SRS for the maximally exposed offsite individual and the general population residing within the 80 kilometer (50 mile) radius surrounding Building 105-L are presented in **Table E–15**. For analysis purposes, the released tritium was assumed to be in oxide form.

Table E–15 Annual Radiological Impacts to the Public From Normal Operational Releases During Melt and Dilute Processing at Building 105-L Under Alternative 5 at SRS

Population		*Maximally Exposed Offsite Individual*	
Collective Dose (person-rem)	***Latent Cancer Fatalities***	***Dose (millirem)***	***Latent Cancer Fatality Risk***
0.0076	3.8×10^{-6}	0.00010	5.0×10^{-11}

Worker Impacts

Worker population and worker impact doses associated with melt and dilute processing at SRS were based on 100 workers and the site administrative dose limit of 500 millirem per year for each worker and are consistent with those presented in the SRS Spent Nuclear Fuel Management Final EIS (DOE 2000). The SRS radiation control program incorporates the DOE Administrative Control Level of 2,000 millirem per year per person established for all DOE activities in DOE Order N441.1. Doses and associated impacts are based on a three-year processing period. **Table E–16** presents estimated values to both the average worker and entire work force population.

Table E–16 Annual and Cumulative Worker Radiological Impacts From Normal Operational Activities During Melt and Dilute Operations at Building 105-L Under Alternative 5 at SRS

Worker Population		*Individual Worker*	
Collective Dose (person-rem per year)	***Latent Cancer Fatalities From Three Years Melt and Dilute Processing***	***Individual Dose (millirem per year)***	***Latent Cancer Fatality Risk From Three Years Melt and Dilute Processing***
50	0.060	500 [a]	0.00060

[a] 500 millirem per year is the site annual ALARA administrative dose limit at SRS. The average worker dose is about 50 millirem per year (DOE 2000).

E.4.5 Melt and Dilute Processing at ANL-W (Alternative 6)

In Alternative 6, the blanket and driver spent nuclear fuel elements would need to be prepared at the ANL-W facilities prior to their processing at ANL-W. Preparation of the fuel at ANL-W for the melt and dilute process requires only that the fuel be cleaned to remove sodium prior to melt and dilute processing; decladding of the blanket and driver spent nuclear fuel is not necessary. This activity would be performed in the argon cell of the Hot Fuel Examination Facility. The following discussion addresses the radiological impacts of normal operations at ANL-W for the preparation and melt and dilute treatment of the blanket and driver spent nuclear fuel.

Gaseous Emissions

Fuel preparation would occur at the Hot Fuel Examination Facility. These activities would cause gaseous fission products to be released into the argon cell. As stated earlier in Section E.4.1, krypton-85 and elemental tritium (H-3) are the most prevalent gaseous radionuclides that would be released to the

environment. The tritium released into the cell would not be oxidized because of a very low presence of oxygen and humidity in the argon cell. The argon cell also contains an equilibrium concentration of other radionuclide isotopes. Section E.4.1 provides a list of various isotopes that are present in the argon cell in nanocuries (10^{-9} curies) and are released to the atmosphere through the facility stack, along with the krypton-85 and elemental tritium.

The melt and dilute alternative consists of two distinct operations. The spent nuclear fuel first would be declad and cleaned to remove as much sodium as possible and then would be treated using melt and dilute process. Decladding and cleaning operations may start as early as 2003 and could continue for six years. The melt and dilute treatment would be expected to begin two years after the start of the decladding and cleaning operations. The gaseous fission products (including tritium and krypton) would be released during the decladding and cleaning of the spent nuclear fuel, when the fuel temperature would be raised to approximately 500 °C (930 °F). During the melt and dilute process itself, additional radionuclides would be volatilized and particulates would be released. The volatilized elements would be condensed and collected while the airborne particulates would be filtered through a filtration system that reduces any release by a factor of at least 0.00001. Analysis performed for the evaluation of normal operations for Alternative 1, Section E.4.1, showed that, for similar conditions, over 99 percent of the population and maximally exposed offsite individual doses would come from the release of tritium and krypton from processing both the blanket and driver spent nuclear fuel. Therefore, the doses from the release of tritium and krypton are used to represent the offsite impacts of normal operation releases. These releases would occur during the six years of decladding and cleaning activities, beginning in approximately 2003.

The maximum released curies of radioactive gaseous emissions would occur when preparation of the blanket and driver spent nuclear fuel is performed simultaneously. This simultaneous operation was estimated to occur over a six-year period starting in 2003. Based on a blanket spent nuclear fuel preparation throughput of 10 metric tons of heavy metal and a driver spent nuclear fuel process rate of about 1.7 metric tons of heavy metal annually, about 2,162 curies of elemental tritium and 32,650 curies of krypton-85 would be released to the atmosphere annually (see **Table E-17**). This release rate would last about two years, or until all of the EBR-II blanket spent nuclear fuel and the driver spent nuclear fuel assemblies were processed. Afterward the release rate would drop during the processing of the Fermi-1 blanket spent nuclear fuel (the release rate for the processing of 10 metric tons of heavy metal of Fermi-1 blanket spent nuclear fuel is less than 1 curie of elemental tritium and 6.6 curies of krypton-85).

Table E–17 Maximum Annual Radiological Gaseous Emissions During Melt and Dilute Operations Under Alternative 6 at ANL-W

Spent Nuclear Fuel Type	*Facility*	*Maximum Processing Rate (metric tons of heavy metal per year)*	*Duration (years)*	*Annual Release (curies)*	
				Tritium	*Krypton-85*
Driver fuel	Hot Fuel Examination Facility	1.7	2	2091	32,130
EBR-II blanket fuel	Hot Fuel Examination Facility	10	2.4	71.2	520
Fermi-1 blanket fuel	Hot Fuel Examination Facility	10	3.6	0.76	6.6
Maximum annual release [a]			2	2,162	32,650

[a] Maximum annual release rate applies to the two years during which both EBR-II driver and blanket spent nuclear fuel are processed.

Population Impacts

The maximum annual doses to the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding ANL-W are presented in **Table E–18**. As stated in

Section E.4.1, the dose resulting from the release of tritium is highly dependent upon its chemical form. The doses presented in Table E–18 result from releases that are assumed to be 1 percent oxidized tritium, the same assumption used in the analysis of Alternative 1. These impacts are calculated for the preparation and processing of the sodium-bonded blanket and driver spent nuclear fuel assemblies and the total maximum impacts. During the four-year period when only EBR-II or Fermi-1 blanket spent nuclear fuel is being processed, the doses and latent cancer fatality risk would be smaller than the total presented in the table, reduced in direct proportion to the amount of material released.

Table E–18 Annual and Total Radiological Impacts to the Public From Operational Releases Under Alternative 6 at ANL-W

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Processing	***Spent Nuclear Fuel Type***	***Collective Dose (person-rem per year)***	***Latent Cancer Fatalities (number of cancers)***	***Dose (millirem per year)***	***Latent Cancer Fatality Risk***
1 to 2	Driver	0.012	6.0×10^{-6}	0.0020	1.0×10^{-9}
	EBR-II blanket	0.00028	1.4×10^{-7}	0.000048	2.4×10^{-11}
	All fuel	0.012	6.1×10^{-6}	0.0020	1.0×10^{-9}
3	EBR-II blanket	0.00011	5.5×10^{-8}	0.000019	9.5×10^{-12}
	Fermi-1 blanket	1.9×10^{-6}	9.5×10^{-10}	3.3×10^{-7}	1.6×10^{-13}
	All fuel	0.00012	5.6×10^{-8}	0.000019	9.7×10^{-12}
4 to 6	Fermi-1 blanket	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
	All fuel	3.2×10^{-6}	1.6×10^{-9}	5.5×10^{-7}	2.7×10^{-13}
Project total [a]		0.024	0.000012	0.0040	2.0×10^{-9}

[a] Maximum annual radiological impacts occur during concurrent processing of EBR-II driver and blanket spent nuclear fuel.

It should be noted that the radiological impacts presented in Table E–18 are based on the assumption of simultaneous operation of blanket and driver fuel. If the fuel preparation were to be performed along with the melt and dilute process for each fuel type separately, then the emissions would occur over a 10-year period starting in 2005. This would result in a lower annual dose to the public over a longer duration leading to the same project total dose as presented in Table E–18.

Worker Impacts:

Due to the uncertainties in the start of operation, for the purposes of analysis in this EIS, DOE assumes that the fuel preparation would start at 2003 and the melt and dilute processing of all fuel would end in 2015, for a total of 12 years of operations. The worker activities during fuel preparation and melt and dilute process would be similar to the activities for Alternatives 1 through 5 at ANL-W. Therefore, the annual worker population dose and average worker dose would be 22 person-rem and 60 millirem, respectively. If these estimates were extended over 12 years of treatment activities plus one year for the deactivation of the facility (with 33-person-rem of dose), the project total worker population dose would be 297 person-rem, leading to a risk of 0.12 latent cancer fatalities.

E.4.6 Storage/Direct Disposal (No Action Alternative)

In the No Action Alternative, the blanket and driver spent nuclear fuel assemblies would remain in their current storage facilities at ANL-W and Idaho Nuclear Technology and Engineering Center (INTEC) until a future disposal option is selected. Potentially, the spent nuclear fuel could remain in its current location until its preparation for disposition sometime before the end of 2035. (All of the sodium-bonded spent nuclear fuel must be removed from the site and moved out of the State of Idaho to fulfill the requirements

of the DOE-State of Idaho Settlement Agreement and Consent Order [see Appendix K].) The only activities associated with continued storage, other than monitoring, would be the repackaging of 5 metric tons of blanket spent nuclear fuel (over a two-year period ending in 2001); the repackaging of spent nuclear fuel found to have degraded (e.g., leaking fuel and storage canister); the transfer of 1.2 metric tons of driver spent nuclear fuel currently in wet storage at Basin 603 to dry storage; and the repackaging of all of the spent nuclear fuel prior to its removal from the State of Idaho. The following discussion addresses the radiological impacts of normal operations at ANL-W and INTEC for the continued storage of the blanket and driver spent nuclear fuel.

Gaseous Emissions:

Under both options in this alternative potential radiological releases from sodium-bonded spent nuclear fuel are very small. Under either option, the sodium-bonded spent fuel would remain in storage in sealed canisters while at INEEL (i.e., INTEC or ANL-W). This fuel needs to be removed from the State of Idaho by January 1, 2035, consistent with the DOE-State of Idaho Settlement Agreement and Consent Order, (see Appendix K). However, degradation of sodium-bonded spent nuclear fuel or its enclosure (e.g., a sealed canister) during storage cannot be ruled out. It is expected that a percentage (a small fraction) of the fuel would be degraded during storage, allowing its gaseous fission products to enter the storage canister. These fission gases would be released to the environment only if the sealed canister were to fail or be opened during fuel handling for examination and repackaging.

The current experience of sodium-bonded spent nuclear fuel storage at INTEC (Basins 603 and 666, wet storage facilities) and ANL-W (Radioactive Scrap and Waste Facility, a dry storage facility) indicates some small fuel degradation problems during the storage period (ANL 2000). For example, during the Electrometallurgical Treatment Research and Demonstration Project, only one canister was observed to have a degraded driver fuel element among the 0.4 metric tons of heavy metal (100 assemblies) of EBR-II driver spent nuclear fuel treated. The degraded fuel was among 6,100 fuel elements that were in dry storage for an average of about four years. Based on this limited experience, the likelihood of fuel degradation during dry storage for the driver spent nuclear fuel would be about 0.005 percent per year. All fuel stored at the Radioactive Scrap and Waste Facility is in cathodically protected liners. At ANL-W, no failures of cathodically protected liners have occurred; therefore, any fuel degradation while in storage is not expected to lead to immediate releases into the atmosphere. Since the driver spent nuclear fuel that failed was in storage for a short period, the failure rate was adjusted to 0.015 percent per year, considering an error factor of 3 (or an uncertainty factor of 10). Therefore, if the fuel were to remain in dry storage for 35 years, about 0.5 percent of the sodium-bonded driver spent nuclear fuel would be in a degraded condition.

The EBR-II fuel at INTEC's Basins 603 and 666 are stored inside stainless steel sealed cans to prevent the contact of basin water with the fuel cladding. The experience at INTEC indicates a higher likelihood of fuel/can degradation in wet storage. A total of 3,624 fuel cans of spent nuclear driver fuel currently is stored at INTEC's Basins 603 and 666. There were 2,148 cans in Basin 603, with an average storage of about 17 years. During this period, 10 cans have shown degradation and water in-leakage, leading to an estimated fuel can failure rate of about 0.03 percent per year. This failure rate was adjusted to 0.10 percent per year, consistent with the assumption made for the driver spent nuclear fuel. Water in-leakage had caused fuel degradation and hydrogen generation from sodium water reactions. The sodium-bonded spent nuclear fuel in Basin 666 has been in storage for about 12 years on average with no observed fuel can failure. All spent nuclear fuel, including the sodium-bonded spent nuclear fuel currently in Basin 603 at INTEC, is to be transferred to dry storage by December 2000, independent of the actions considered in this EIS. During transfer, each fuel can containing sodium-bonded spent nuclear fuel would be examined for water in-leakage. If a fuel can were found to be degraded (containing water), it would be packaged and sent to ANL-W for further examination and repackaging. After transfer to a dry storage facility at INTEC, the likelihood of fuel

degradation would be similar to that at ANL-W. The sodium-bonded spent nuclear fuel in Basin 666 would remain in (wet) storage until the planned defueling and facility closure in the year 2023.

In the basin (wet storage), fuel can degradation would cause a fuel-water reaction, producing hydrogen gas, which would create bubbles in the basin leading to failure detection. Upon detection, the fuel can would be removed and sent to ANL-W for further examination and repackaging. If no action were taken to treat the sodium-bonded spent nuclear fuel by 2023, the fuel would be removed from the basin and placed in storage or repackaged in preparation for removal from the State of Idaho by 2035.

Based on the above experience, the likelihood of fuel failure during dry storage was estimated to be about 0.015 percent per year. When the fuel is in dry storage, fuel degradation would not be detected. Over a storage period of up to 2035, it was estimated conservatively that about 0.5 percent of the fuel would be in a degraded condition. This estimate also would be used for the blanket spent nuclear fuel, even though no blanket spent nuclear fuel element failures during storage have been observed. The likelihood of fuel/can failure in wet storage is about 0.10 percent per year. Therefore, for consistency with the assumption of driver spent nuclear fuel failure in dry storage, it was assumed that about 3 percent of the spent nuclear fuel would have failed during the wet storage period of up to 2035.

Using the above spent nuclear fuel failure assumptions, the estimated radiological gaseous emissions during each option of the No Action Alternative are summarized below.

- *Continued Storage Option*—Under this option, only 107 cans containing about 5 metric tons of heavy metal of blanket spent nuclear fuel at ANL-W would be repackaged and returned to dry storage within the first two years. This would lead to a release of about 0.04 curies of tritium and 0.3 curies of krypton-85 over the first two years. Over the same period, the releases of other gaseous fission products, such as iodine-129, would be less than 10^{-7} curies. The spent nuclear fuel in INTEC's Basin 666, which was assumed to remain in the pool up to 2035, would release about 1 curie of tritium oxide (the elemental tritium in the fuel was assumed to be oxidized in the water), 15.1 curies of krypton-85, and 1.5×10^{-6} curies of iodine-129 annually. At some future time, all sodium-bonded spent nuclear fuel at ANL-W would have to be repackaged in preparation for transferring out of the State of Idaho by 2035, consistent with the DOE-State of Idaho Settlement Agreement and Consent Order. The spent nuclear fuel in dry storage at ANL-W and INTEC would release 16.91 curies of tritium oxide, 254.1 curies of krypton-85, and 0.000011 of iodine-129 during fuel repackaging for removal which would occur over three years. The total radiological releases over 35 years would be: 50.51 curies of tritium oxides, 760.3 curies of krypton-85, and 0.000018 curies of iodine-129. Due to uncertainties about when the repackaging would occur, the blanket and driver spent nuclear fuel radionuclide inventories were not decayed beyond the 2000 calendar year. This could result in overestimating the gaseous tritium content by a factor of 8.

- *Direct disposal*—Under this option, all the sodium-bonded spent nuclear fuel at INTEC or ANL-W would be repackaged at ANL-W. The activities to repackage the sodium-bonded fuel in high-integrity cans would occur over three years. These activities would occur sometime after those performed in the first two years under the storage option of the No Action Alternative and before January 2035, the target date for removal of spent nuclear fuel from the State of Idaho. The fuel currently at INTEC would be transferred to ANL-W between 2003 and 2023. The 2023 date corresponds to the target date for closure of the facility containing Basin 666. Therefore, similar to the previous option, the releases under this option would occur over two distinct periods: (1) over two years during repackaging of the blanket spent nuclear fuel (see continued storage option above); and (2) over three years during repackaging and preparation for direct disposal. Since similar activities are performed under both options, the total radiological releases also would be similar; that is, about 51 curies of tritium, 760 curies of krypton-85, and 0.00002 curies of iodine-129 over the entire period.

Population Impacts:

The doses to the maximally exposed offsite individual and the general public residing within the 80 kilometer (50 mile) radius surrounding the ANL-W site are presented in **Table E–19.** All releases producing these impacts are modeled as originating from ANL-W. Due to the relative locations of ANL-W, INTEC, and the surrounding population, the impacts of releases from INTEC are bounded by the impacts of releases from ANL-W. The dose resulting from the release of tritium is highly dependent upon its chemical form. The doses in Table E–19 result from releases that are assumed to be oxidized tritium.

Table E–19 presents the radiological impacts for the storage option as described above (impacts from the direct disposal option would be similar), and includes contributions from the following releases:

- The repackaging of 5 metric tons of blanket spent nuclear fuel during the first two years,
- Leakage from 2 metric tons of driver spent nuclear fuel in wet storage for one year,
- Leakage from 0.8 metric tons of driver spent nuclear fuel in wet storage (Basin 666) for 31 years (1.2 metric tons would be moved from wet [Basin 603] to dry storage within the first year), and
- The repackaging of all the stored sodium-bonded spent nuclear fuel during the final three years.

Table E–19 Annual and Total Radiological Impacts to the Public From Normal Operations Under the No Action Alternative

		Population		*Maximally Exposed Offsite Individual*	
Year(s) of Storage	***Spent Nuclear Fuel Type***	***Collective Dose (person-rem per year)***	***Latent Cancer Fatalities (number of cancers)***	***Dose (millirem per year)***	***Latent Cancer Fatality Risk***
1	Driver	0.00063	3.2×10^{-7}	0.00011	5.5×10^{-11}
	Blanket	4.8×10^{-6}	2.4×10^{-9}	8.3×10^{-7}	4.1×10^{-13}
	All fuel	0.00064	3.2×10^{-7}	0.00011	5.6×10^{-11}
2	Driver	0.00025	1.3×10^{-7}	0.000044	2.2×10^{-11}
	Blanket	4.8×10^{-6}	2.4×10^{-9}	8.3×10^{-7}	4.1×10^{-13}
	All fuel	0.00026	1.3×10^{-7}	0.000045	2.3×10^{-11}
3 to 32	Driver	0.00025	1.3×10^{-7}	0.000044	2.2×10^{-11}
	Blanket	0	0	0	0
	All fuel	0.00025	1.3×10^{-7}	0.000044	2.2×10^{-11}
33 to 35	Driver	0.0014	7.0×10^{-7}	0.00024	1.2×10^{-10}
	Blanket	0.000072	3.6×10^{-8}	0.000013	6.3×10^{-12}
	All fuel	0.0015	7.5×10^{-7}	0.00026	1.3×10^{-10}
Project total [a]		0.013	6.5×10^{-6}	0.0023	1.1×10^{-9}

[a] Annual maximum occur during repackaging of spent nuclear fuel in preparation for shipping off site.

Worker Impacts:

The worker activities under the No Action Alternative during spent nuclear fuel repackaging would be similar to some of the activities performed under Alternative 1. Therefore, for the five years that repackaging activities are ongoing, the first two years and the last three years that the fuel remains on site, the annual worker dose and the worker population dose would be bounded by those values provided in Section E.4.1.

E.5 IMPACTS OF EXPOSURES TO HAZARDOUS CHEMICALS ON HUMAN HEALTH

The potential impacts of exposure to hazardous chemicals released to the atmosphere were evaluated for routine operations associated with the alternatives analyzed in this EIS. The public residing at the site boundary was the receptor considered in this evaluation. Health impacts to workers from hazardous chemicals were not evaluated quantitatively because of the use of personal protective equipment and engineering process controls. Their exposure is limited to levels within applicable Occupational Safety and Health Administration Permissible Exposure Limits, or the American Conference of Governmental Industrial Hygienists Threshold Limit Values.

Human health effects could result from exposure to hazardous/toxic chemicals through one or more of the three common pathways: inhalation, ingestion, and/or dermal (skin) contact. The effects from a particular pathway will depend essentially on the properties of the toxic chemical, its concentration in one or more environmental media (air, water, and soil), and human behavior. Exposure may be dominated by contacts with chemicals in a single medium or may reflect concurrent contacts with multiple media. Therefore, the exposure assessment provides an estimate of how chemicals travel to a receptor, and how those chemicals come into contact with the receptor's body. It also determines whether the chemicals present in the environmental medium are of sufficient concentration to cause significant adverse effects. The exposure assessment assumes inhalation to be the only pathway and air the only medium. This simplification is based principally on the volatility of the chemicals released. Normal human behavior also is considered (i.e., an individual is assumed to perform activities under normal conditions). To maximize the impact of the exposure, the analysis also assumes that the released chemicals will remain in the air with no or negligible partitioning to other media (i.e., water and ground). So no dermal contact or ingestion is considered in this assessment.

Hazardous chemical releases from routine operations generally are expected to result in concentrations below levels that would cause acute toxic health effects. Acute toxic health effects generally result from short-term exposure to relatively high concentrations of the toxic contaminant, such as those resulting from accidental releases. Long-term exposures to lower concentrations can produce adverse chronic health effects, both carcinogenic and noncarcinogenic. Excess incidences of cancer are the endpoint of carcinogenic effects. However, a spectrum of chemical-specific noncancer health effects (e.g., headaches, skin irritation, neurotoxicity, immunotoxicity, reproductive and genetic toxicity, liver/kidney toxicity, and developmental toxicity) could be observed for noncarcinogenic compounds.

E.5.1 Methodology

This EIS estimates the noncancer health effects by comparing the annual concentrations of contaminants to the Reference Concentrations published in the Integrated Risk Information System (EPA 1999). The potential toxic effects on an individual from exposure to a toxic chemical are evaluated by dividing the estimated inhalation concentration of that chemical by its Reference Concentration value to obtain a noncancer hazard quotient (EPA 1989). For exposure to multiple compounds, hazard quotients are calculated for each toxic chemical and then are summed to generate a hazard index:

$$HI = \sum_i \frac{CA_i}{RfC_i}$$

where

RfC_i = Reference Concentration for chemical *I* (in micrograms per cubic meter)
CA_i = Concentration of the chemical *I* in the air (in micrograms per cubic meter)
HI = Hazard Index

The hazard index is the estimate of the total noncancer toxicity impact. According to the EPA risk assessment guidelines, if the hazard index value is less than or equal to 1, the exposure is unlikely to produce adverse toxic effects. However, if it exceeds 1, adverse toxic effects may result from exposure to the considered chemicals.

The risks from exposure to carcinogenic chemicals are evaluated using chemical-specific unit risk factors, which are the estimates of the upper-bound lifetime probability of an individual developing cancer from exposure to the chemical and the chemical concentration in the air. The unit risk factors for carcinogenic chemicals are provided in the EPA's Integrated Risk Information System database. Therefore, for carcinogenic chemicals, the risk is estimated by the following equation (EPA 1989):

$$Risk = 1 - \exp [- CA \times URF]$$

where

CA = Contaminant concentration in the air (in micrograms per cubic meter)
URF = Unit risk factor for inhalation specific to the contaminant obtained from the Integrated Risk Information System in units of cancer per micrograms per cubic meter

Since the value in the bracket is generally small (less than 0.01), the equation is simplified to:

$$Risk = CA \times URF$$

E.5.2 Assumptions

The air is assumed to be the principal medium by which an individual would be exposed to released hazardous chemicals, and the health effects are calculated based solely on inhalation pathway. In addition, no synergistic or antagonistic effects are assumed to occur from the exposure to multiple hazardous chemicals.

Cancer risks associated with exposure to carcinogenic chemicals were not summed to provide a single cancer risk value. In terms of risk evaluation, a value integrated over multiple chemicals is not always appropriate. One cannot simply add the risk values of individual chemicals to calculate the overall risk. With the risk assessment guidelines and the weight of evidence (EPA 1999), a new approach to carcinogenic risk characterization is being implemented. Thus, even though several chemicals may be shown to induce cancer, they do not necessarily act on the same organ. For example, benzene and formaldehyde are both carcinogenic. Formaldehyde could induce nasal cancer (Andjelkovitch, et. al. 1995), while benzene could cause leukemia. Thus, their residual cancer risk is not cumulative, and the cancer risk for each carcinogenic chemical would be presented separately.

E.5.3 Hazardous Chemical Releases to the Environment and Associated Impacts

This section summarizes the estimated hazardous chemical releases to the environment as well as resulting impacts associated with various alternatives assessed in this EIS.

E.5.3.1 Hazardous Chemical Impacts at ANL-W (All Alternatives)

Under all alternatives, including No Action, small quantities of hazardous chemicals are generated from the operation of the emergency diesel generators supporting both the Fuel Conditioning Facility and Hot Fuel Examination Facility at ANL-W. The emissions from these are independent of any of the processes addressed in this EIS. The released chemicals include acetaldehyde, acrolein, benzene, butadiene, formaldehyde, and toluene. The emissions from these diesel generators were modeled as a volume source

releasing at ground level. In addition, the electrometallurgical treatment of driver spent nuclear fuel under Alternatives 1 through 5 releases small quantities of cadmium. This release would occur as an elevated release (61 meters [200 feet]) from the Fuel Conditioning Facility stack.

Site boundary hazardous chemical concentrations in the atmosphere from releases at ANL-W were estimated using the SCREEN 3 computer program (Version 96043), an EPA-approved worst-case screening model (EPA 1995). The model predicts 1-hour concentrations at the site boundary based on a set of worst-case meteorological conditions. Concentrations were predicted at 16 sectors along the site boundary, assuming a flat terrain. The maximum 1-hour concentration at the site boundary then was selected for the determination of health effects. This concentration was converted to an annual concentration using a regulatory-approved scaling factor of 0.05 (SCDHEC 1993). **Table E–20** summarizes the results. These results indicate that no adverse toxic health effects and cancer potency are expected from exposure to hazardous chemical releases under all alternatives at ANL-W.

Table E–20 Hazardous Chemical Impacts to the Public From Operational Activities at ANL-W for All Alternatives Including No Action

Chemical	*Modeled Emission Rate (grams per second)*	*Annual Concentration (milligrams per cubic meter)*	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligram per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
1,3-Butadiene	0.000012	3.6×10^{-8}	None	0.28	None	9.9×10^{-9}
Acetaldehyde	7.6×10^{-6}	2.3×10^{-8}	0.009	0.0022	2.5×10^{-6}	5.0×10^{-11}
Acrolein	2.4×10^{-6}	7.1×10^{-9}	0.00002	None	0.00035	None
Cadmium	1.5×10^{-10}	3.6×10^{-13}	None	1.8	None	6.5×10^{-13}
Benzene	0.00023	6.9×10^{-7}	None	0.0078	None	5.4×10^{-9}
Formaldehyde	0.000024	7.1×10^{-8}	None	0.013	None	9.2×10^{-10}
Toluene	0.000083	2.5×10^{-7}	0.4	None	6.2×10^{-7}	None
				Hazard Index	0.000353	N/A

N/A = Not applicable.
Source: EPA 1999, modeling results.

E.5.3.2 Hazardous Chemical Impacts at SRS (Alternatives 3 and 5)

Hazardous chemical releases associated with the PUREX and melt and dilute processes at SRS were estimated based on information provided in the SRS Spent Fuel Management Final EIS (DOE 2000). The hazardous chemical release estimates at SRS were essentially independent of the processes evaluated; the chemicals are generated from operation of supporting facilities and equipment (i.e., emergency diesel generator, site-wide powerhouse coal-fired boilers and fuel-oil steam generated boilers). The hazardous chemical release values selected for this EIS were the SRS estimated values that were released during treatment of about 20 metric tons of heavy metal of declad and cleaned EBR-II blanket spent nuclear fuel, similar to the fuel considered for treatment at SRS under the SBSNF EIS. These SRS values were adjusted to account for the mass of spent nuclear fuel being treated (about 57 metric tons of heavy metal) at SRS under Alternatives 3 and 5. In addition, the annual hazardous chemical concentrations were estimated using the 24-hour concentration values given in the SRS EIS and the regulatory-approved scaling factor of 0.125 to convert the 24-hour concentration to an annual concentration (SCDHEC 1993).

Tables E–21 and **E–22** present the results of the hazardous chemical analyses for Alternatives 3 and 5, respectively. These results indicate that no adverse toxic health effects and cancer potency are expected from exposure to hazardous chemical releases under these alternatives at SRS.

Table E–21 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 3 at SRS

Chemical	*Annual Concentration (milligrams per cubic meter)* [a]	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligrams per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	1.4×10^{-6}	None	0.0078	None	1.1×10^{-8}
Ethyl benzene	1.3×10^{-6}	1	None	1.3×10^{-6}	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	1.4×10^{-6}	0.2	None	7.1×10^{-6}	None
Manganese	1.3×10^{-6}	0.000050	None	0.025	None
Methyl ethyl ketone	2.5×10^{-6}	1	None	2.5×10^{-6}	None
Methylene chloride	7.1×10^{-7}	None	0.00047	None	3.3×10^{-10}
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	1.4×10^{-6}	0.4	None	3.5×10^{-6}	None
Vinyl acetate	1.3×10^{-6}	0.2	None	6.3×10^{-6}	None
			Hazard Index	0.025	N/A

N/A = Not applicable.
[a] These concentrations were estimated based on values given in Bickford et al. 1997.
Source: EPA 1999.

Table E–22 Hazardous Chemical Impacts to the Public From Operational Activities Under Alternative 5 at SRS

Chemical	*Annual Concentration (milligrams per cubic meter)* [a]	*Reference Concentration Inhalation (milligrams per cubic meter)*	*Unit Cancer Risk (risk per milligrams per cubic meter)*	*Hazard Quotient*	*Cancer Risk*
Benzene	ND	None	0.0078	None	ND
Ethyl benzene	ND	1	None	ND	None
Formaldehyde	1.3×10^{-6}	None	0.013	None	1.6×10^{-8}
Hexane	1.3×10^{-6}	0.2	None	6.3×10^{-6}	None
Manganese	ND	0.00005	None	ND	None
Methyl ethyl ketone	1.3×10^{-6}	1	None	1.3×10^{-6}	None
Methylene chloride	ND	None	0.00047	None	ND
Naphthalene	1.3×10^{-6}	0.003	None	0.00042	None
Toluene	1.3×10^{-6}	0.4	None	3.1×10^{-6}	None
Vinyl acetate	ND	0.2	None	ND	None
			Hazard Index	0.00043	N/A

N/A = Not applicable, ND = Not detectable.
[a] These concentrations were estimated based on values given in Bickford et al. 1997.
Source: EPA 1999.

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